

SAFETY EVALUATION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
RELATED TO AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. DPR-6
CONSUMERS ENERGY
BIG ROCK POINT NUCLEAR PLANT
DOCKET NO. 50-155

1.0 INTRODUCTION

By letter dated April 1, 2003 (Ref. 1), and supplemented by letter dated July 1, 2004 (Ref. 2), Consumers Energy (CE or the licensee) submitted a request to amend Facility Operating License No. DPR-6 for the Big Rock Point Nuclear Plant (BRP or the facility). In accordance with the requirements of CFR 50.82(a)(9) (Ref. 3) the licensee submitted a license termination plan for its facility to the U.S. Nuclear Regulatory Commission (NRC). Under the provisions of 10 CFR 50.82(a)(10), the NRC approves license termination plans by license amendment. Thus, the licensee has requested the addition of a new License Condition to the BRP License. The new license condition would incorporate the NRC approved License Termination Plan (LTP) into the BRP license, and allow the licensee to make certain changes to this approved LTP without prior NRC review or approval. The Big Rock Point License Termination Plan describes an acceptable approach for demonstrating compliance with the radiological criteria for unrestricted use, as defined by 10 CFR 20.1402 (Ref. 4), by reducing residual radioactivity to as low as reasonably achievable (ALARA) and meeting a site release criterion of 25 millirem TEDE per year from all pathways. This is done by using appropriate dose modeling methods, pathways and parameters, and acceptable final radiation survey methods.

The licensee will implement all the provisions of the License Termination Plan submitted on April 1, 2003, with a supplemental revision on July 1, 2004, as approved in the SER dated March 24, 2005. The licensee may make changes to the License Termination Plan without prior approval, provided the proposed changes do not:

- (i) Require Commission approval pursuant to 10 CFR 50.59;
- (ii) Violate the requirements of 10 CFR 50.82(a)(6), 10 CFR 50.82(a)(7), or 10 CFR 72.48;
- (iii) Reduce the coverage requirements for scan measurements;
- (iv) Change the statistical test to other than the Sign Test for data evaluation;
- (v) Increase the radionuclide-specific derived concentration guideline limits (DCGLs) or area factors;

- (vi) Increase the radioactivity level, relative to the applicable DCGL, at which an investigation occurs;
- (vii) Changes the classification of a survey unit from a more restrictive classification to a less restrictive classification (e.g., Class I to Class 2). Definitions for the different classifications for surface soils are provided in Chapter 5 the LTP; or
- (viii) Increase the probability of making a Type I (α) decision error.
- (ix) The licensee will submit an updated Final Hazards Summary Report as specified in 10 CFR 50.71 (e).

In addition, the licensee will certify in its application for Part 50 license termination that it has met the radiological criteria for unrestricted use, as defined by 10 CFR 20.1402, by meeting a site release criterion of 25 millirem TEDE per year from all appropriate pathways, and that residual radioactivity has been reduced to levels that are ALARA, in accordance with the approved License Termination Plan. The licensee will also request NRC to confirm this certification.

2.0 EVALUATION

CE submitted its LTP on April 1, 2003 (Ref. 1), with a supplemental revision dated July 1, 2004 (Ref. 2), in accordance with 10 CFR 50.82(a)(9), that requires the LTP to contain the following information: (1) a site characterization; (2) identification of remaining dismantlement activities; (3) plans for site remediation; (4) detailed plans for the conduct of final radiation survey; (5) a description of the end use of the site, if a restricted option is selected; (6) an updated site-specific estimate of remaining decommissioning costs; and (7) a supplement to the environmental report, pursuant to 10 CFR 51.53 (Ref. 5) describing any new information or significant environmental changes associated with the licensee's proposed termination activities.

The LTP describes BRP's approach for demonstrating compliance with radiological criteria for unrestricted use. In addition, the licensee requested the authority to make certain changes to the LTP, once approved by NRC.

2.1 Site Characterization

Site characterization surveys are conducted to determine the nature and extent of radioactive contamination in buildings, plant systems and components, site grounds, and surface and ground water. The major objectives of characterization activities are to: permit the planning and conduct of remediation activities; confirm the effectiveness of remediation methods; provide information to develop specifications for final status surveys (FSSs); define site and building areas as survey units and assign survey unit classifications; and provide information for dose modeling.

CE conducted site characterization activities by a historical site assessment (HSA), a scoping survey, and a characterization survey. It conducted the scoping survey in 1993-1994. Site

characterization survey activities included the collection of various types of samples, including soil, sediment, water, concrete, and metal. Surveys and sampling conducted during site characterization are based on biased and judgmental measurements. In accordance with 10 CFR 50.82(a)(9)(ii)(A), CE provides radiological site characterization of the site in Chapter 2 of the LTP. They provide the results of sample analyses, and the use of the results in identifying the significant radionuclides expected to be present after remediation, in LTP Chapter 2, Appendix 2-E.

In support of characterization efforts, the licensee conducted an HSA. The HSA used information from decommissioning records; employee interviews; Health Physics Logbook; corrective action records; previous decommissioning studies; waste shipment records; operational survey records; and annual radiological environmental reports. The HSA process identified 63 events, during the operational life of the plant, with known or potential radiological impact on the environment. CE used the results of the HSA to guide remediation activities and to confirm the appropriateness of the radiological source terms used for the dose model, as more site information is being collected.

The licensee conducted a series of sample analyses using site media believed to represent the distribution of radionuclide contaminants, and their decay-corrected distribution, over the operational history of the plant. Tables 2-11, 2-13, and 2-14 of the LTP identify 24 radionuclides present at the site: H-3; C-14; Mn-54; Fe-55; Ni-59; Co-60; Ni-63; Zn-65; Sr-90; Tc-99; Ag-110m; I-129; Cs-134; Cs-137; Eu-152; Eu-154; Eu-155; Pu-238; Pu-239/240; Pu-241; Am-241; and Cm-243/244. These radionuclides include fission and activation products, which are typical of those found in boiling water reactor plants and are similar to those radionuclides described in NUREG/CR-3474 (Ref. 6), and NUREG/CR-4289 (Ref. 7). Based on dose model assumptions and expected time at which it will remediate the site, the licensee identified, in Table 6-2 of the LTP, the following radionuclides as contributing to the dose after license termination: H-3; Mn-54; Fe-55; Co-60; Sr-90; Cs-137; Eu-152; Eu-154; and Eu-155. Accordingly, these radionuclides will form the basis in planning and conducting all FSSs, and demonstrating compliance with the site release criteria. The LTP allows for taking and analyzing additional samples as decommissioning activities warrant.

The licensee applied the guidance in the Draft Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Ref. 8) at the time that they conducted the scoping and characterization surveys. The purpose of these surveys was to assess the extent and magnitude of radioactive contamination levels in site facilities and grounds, and develop the initial decommissioning plans, schedules, and cost estimates.

Site-specific measurements resulting from radiological survey and hydrogeological assessment provided the information necessary to develop finalized DCGL values and radionuclide fractions. The methods and findings for development of final DCGL values are detailed in Chapter 6 of the LTP, Compliance with the Radiological Criteria for License Termination. Typical radionuclide fractions are in Section 2.4.5.4 and analytical results in Tables 2-11 and 2-14.

The types of surveys and sampling methods described for the survey efforts are acceptable. The staff will evaluate survey implementation activities during routine in-process inspections to

ensure that the methodology is adequate to meet the technical objectives of the LTP, and that the results confirm that CE remediated the site and it meets the cleanup criteria.

Quality assurance (QA) and quality control (QC) measures for characterization surveys included requirements for ensuring compliance with 10 CFR Part 50, Appendix B. The QA/QC measures addressed personnel training and qualifications; instrumentation use and control; schedules; procedures; records and document control; audits and surveillance; data collection and verification; sample custody; sample re-analysis; and duplicate and split sample measurements. For samples subjected to laboratory analyses, the QA/QC requirements addressed the qualifications of: laboratories; sample tracking; use of control charts; use of laboratory controls, blanks, and matrix spike samples; daily instrument checks and periodic calibrations; and analytical procedures.

The radionuclides considered important in modeling doses, are presented in the LTP, Section 6.4 and Table 6-2. The dose limit and dose model, and corresponding DCGLs are presented in Section 6.8.2.2 and Tables 6-10 and 6-11 of the LTP.

In March 2003, the licensee completed transferring the spent fuel into concrete dry storage casks located in the newly constructed independent spent fuel storage installation (ISFSI), approximately one half mile south of the industrial area. Both the ISFSI, authorized under the general license provisions of 10 CFR 72.210, and its Security Operations Building will remain active until CE transfers all the spent fuel to the U.S. Department of Energy (DOE), projected by the licensee to occur in calendar year 2012. At that time, it will demolish the ISFSI and Security Operations Building, and survey and release the areas and facilities for unrestricted use in accordance with NRC regulations.

In the approach outlined in the LTP, the licensee proposes to demolish all major structures, including footings. CE will ship the resulting concrete demolition debris off the site for appropriate disposal. It will backfill the remaining below grade excavations with clean material, and all areas will be contoured to local grade elevation. The few facilities that will remain include the road between US-31 and the ISFSI, and the ISFSI and its operations building.

The licensee designated additional areas around the Industrial Area (IA) as potentially impacted by site operations. The Impacted Area (IM) extends over a mile along the Lake Michigan shoreline and contains an area of 0.540 km² (133.3 acres), including the Industrial Area. The majority of the IM is remote from plant operational activities and has little probability of containing residual radioactivity. However, they will survey these areas because of their proximity to the Industrial Area or historical occurrences. An additional small area at the south end of the site, between US-31 and the former railroad spur is classified as impacted due to a small amount of potentially contaminated construction debris moved there; the debris was subsequently removed and the area surveyed. (See LTP Figures 2-2 and 2-6).

The discussion of non-impacted areas (those areas outside the IM) is in section 2.3.3 of the LTP. That portion of the site ranges from low wetlands to mature forested uplands. The Non-Impacted Area is mainly characterized by thick forest and uneven terrain that is generally inaccessible to motorized traffic. The roadway for access to the ISFSI that extends from the industrial area to US-31 will remain until CE removes fuel from the ISFSI. The licensee conducted a radiological survey and concluded that the areas not associated with the ISFSI

were not impacted by plant operations and did not contain radioactivity that was distinguishable from background levels. Chapter 2 and Appendix 2-E present a summary of characterization results and the licensee's evaluation in declaring these areas non-impacted.

The licensee's determination of the initial classifications of known and potentially contaminated areas and structures was based on the HSA, survey results obtained during the scoping survey, routine surveillance, and an integrated evaluation of that information. Section 2.3.3 and Chapter 5 of the LTP, Table 5-2, identify the IMs by survey unit designation, survey unit classification, and approximate size of the areas. LTP Chapter 2, Appendix 2-E, presents more detailed radiological information supporting the initial classifications assigned to site areas.

Section 5.2 of the LTP, particularly Tables 5-2 and 5-3, provides an acceptable process and description for defining IM classifications (Classes 1, 2, and 3) and non-impacted area classification of survey units. LTP Sections 5.3 to 5.6 present the process and criteria that CE will use to plan, design, and implement FSSs taking into account the various types (surface and volumetric) of DCGLs they will apply to open land areas (surface and deep soils), ground water and surface water to demonstrate compliance with the release criteria. LTP Section 6.8 (Table 6-10) presents site-specific DCGLs for surface soil and Table 6-11 presents site-specific DCGLs for subsurface soils. They will conduct the final site surveys using modified guidance from MARSSIM, and NUREG-1507 (Ref. 9). The licensee will use NRC guidance to develop survey procedures to demonstrate compliance with the site-specific criteria for unrestricted release.

The staff finds the site characterization program acceptable, based on the information described in the LTP. NRC will continue to evaluate, by future in-process inspections, the licensee's activities and how this information will be used in implementing the FSS.

2.1.1 Facility Radiological Status

As described in Section 1.3 of the LTP, the plant started operating in September 1962, and was shut down in August 1997. On December 24, 1998, the NRC issued license amendment No. 120 approving a change in the Technical Specifications to reflect the permanently defueled conditions of the plant and operating conditions, and to ensure the long-term safety of the spent fuel.

Significant radiological events are summarized Chapter 2, Appendix 2-B of the LTP. Events reported include spills, both inside buildings and at outdoor locations, soil and sediment contamination from system leaks, and inadvertent relocation of contaminated materials in previously clean areas. A significant event was a 20,000 gallon leak from a piping connection from the Condensate Storage tank in 1984; several occurrences of system leakage were responsible for varying contamination levels in the Reactor Containment and turbine buildings.

In 1998, CE performed a primary system chemical decontamination that reduced primary system piping radioactivity levels by a factor of ten to twenty. The decontamination removed 15 million megaBecquerels (Mbq) [406 Curies (Ci)] of gamma-emitting radionuclides. This resulted in significant dose rate reductions. A noted benefit of the chemical decontamination was the apparent removal of transuranics, which resulted in essentially no alpha contamination

in the primary system piping. The licensee completed the transfer of spent fuel from the reactor fuel pool to the ISFSI in March 2003.

The staff finds the facility radiological status is acceptable.

2.1.1.1 Structures

The only structures that will remain onsite will be those supporting ISFSI operations. Although the licensee expects no buildings utilized during power operation to remain on the site at the time of the FSS, a contingency might arise for a specific structure to be in use at the time of the FSS for the apposite survey unit. If so, CE would survey any such building in accordance with NUREG-1575 guidelines, and include results in the FSSs provided to the NRC. The NRC approved the bulk materials control program for disposing of slightly contaminated concrete and soil at a licensed landfill (Ref. 10). The NRC also approved a revision to the program to allow material that contained PCB's to be disposed at a second landfill licensed to receive that material (Ref. 11). The licensee will transfer all other materials to a licensed radioactive low level waste (LLW) "broker" for packaging and disposal in accordance with its license.

The staff concludes that the approach used by the licensee to characterize demolition debris is acceptable.

2.1.1.2 Systems

CE evaluated each plant system for potentially removable and fixed contamination by various methods, including surveys and sample analyses. As expected, it noted that radioactivity levels were elevated where contamination levels and associated exposure rates were equally high. The licensee is removing all systems and components pursuant to the Post Shutdown Decommissioning Activities Report (PSDAR). At this time, the fuel pool has been drained and cleaned. Because CE is removing and disposing all of this equipment as radioactive waste, the equipment will not contribute to residual contamination levels at the time CE will conduct FSSs. Accordingly, this SER does not discuss the associated radiological properties of plant systems (this information is in LTP Section 2).

The staff concludes that the approach used by the licensee to characterize affected and unaffected systems is acceptable.

2.1.1.3 Activation

CE expects the removal and shipment of neutron-activated reactor components and concrete to eliminate essentially all the activated material from the site.

The staff concludes that the licensee's characterization of neutron-activated material is acceptable.

2.1.1.4 Surface and Deep Soils

The licensee conducted surveys to assess the presence, and extent, of contamination in surface and deep soils, and to identify the scope of remediation necessary to meet the release

criteria for unrestricted site use. It sampled the Non-Impacted Area as part of a regional study to determine the standard background concentration of gamma emitting radionuclides in soil, as summarized in Section 2.3.3 of the LTP. Cs-137 was the only radionuclide (other than naturally occurring radionuclides) identified in soil analyses of Non-Impacted Areas. Cs-137 concentrations in the Non-Impacted Area are less than or equal to those found in a reference area 12 miles south of BRP. The Cs-137 in the Non-Impacted Area is attributable to atmospheric weapons testing and is not of plant origin.

In the IM, subsurface soil samples were collected at locations that the licensee thought were the most representative of contamination levels associated with past spills and leaks. CE conducted the characterization after grouping the plant site into 27 survey packages and groups as described in Section 2.4.5 and Appendix 2-E of the LTP. More than eleven hundred representative samples of soil and groundwater were collected and analyzed from the IM. The primary analytical method was laboratory gamma spectroscopy, in-situ gamma spectroscopy supplemented this. Hard-to-detect radionuclides were evaluated by a combination of laboratory analyses and the surrogate method.

Soil sampling and analyses revealed varying contamination levels, depending on whether plant operations, including spills and leaks had affected the areas. The contaminants found were mainly H-3, Mn-54, Co-60, and Cs-137. CE used soils collected from areas with the highest contamination levels to determine radionuclide distributions, fractions, and surrogate radionuclides for transuranics (TRU) and hard to detect (HTD) nuclides, as shown in Figure 2-12 of the LTP. Of the 389 soil samples analyzed in the most highly contaminated area, 21 were selected for further analysis by an independent laboratory to provide characterization of hard-to-detect radionuclides. The licensee calculated surrogate ratios conservatively. It reported contamination levels of up to 106 pCi/g for Co-60 and up to 31 pCi/g for Cs-137. The results are summarized in LTP Section 2.4.5 and Appendix 2-E.

The staff concludes that the approach used by the licensee to characterize both Impacted and Non-Impacted areas, and the process to identify areas that might require further soil remediation are acceptable. In addition, the licensee committed to remove contaminated surface and subsurface soils and evaluate radionuclide distributions and fractions to confirm that the assumptions used to develop the site dose model and cleanup criteria are still in agreement with site conditions. However, the NRC staff will confirm, during in-process inspections and confirmatory surveys, that residual soil contamination levels in affected and unaffected areas meet the release criteria.

2.1.1.5 Ground Water

The NRC staff evaluated the following: (1) the extent that groundwater at the Big Rock Point (BRP) facility contains plant-generated radionuclides, (2) whether ground water contaminated with plant-generated radionuclides migrated off site, and (3) the impact that this contaminated ground water has upon potential receptors. This evaluation is based upon BRP's LTP and supporting documents and upon the NRC's independent assessment.

The BRP facility comprises approximately 230 hectares (563.6 acres) in Charlevoix, Michigan. The facility occupies approximately one mile along the Lake Michigan shoreline and approximately one mile inland. The non-impacted area consists of 175 hectares (430.4 acres).

The IM, approximately 55 hectares (133.2 acres), occupies the northern portion of the facility along Lake Michigan. Within the impact areas, the Industrial Area, the location of potential radiological concern (Protected Area, Radwaste Compound, and all material routes and storage locations), is an area slightly less than eight hectares (20 acres). Figures 2.1 and 2.2 in the LTP delineate these areas.

BRP's ground water monitoring to the south of the IA within the non-impacted area indicates that plant-generated radionuclides have not impacted the ground water in this area. Also, ground water flows predominately northward toward Lake Michigan; therefore, the likelihood of any plant-generated radionuclides in ground water south of the IA is extremely low.

Based on BRP's HSA and on the ground water flow patterns, the NRC staff concludes that no radiological contamination from plant-generated radionuclides occurred south of the IA. Therefore, the remaining discussion will focus on radiological contamination of the ground water within the IA and other impact areas along Lake Michigan.

Stratigraphy of the Water-Bearing and Associated Units

The BRP site occupies a zone of low topographic relief that parallels the Lake Michigan shoreline. This zone was formerly submerged beneath the ancestral stages of Lake Michigan, and the terrain is characterized by low recessional beach ridges separated by swampy areas. Much of the site is considered wetlands; however, the terrain rises on a till plain to upland drumlins further inland. Elevation at the site gradually rises from 175.9 meters mean sea level (msl) [577 feet] at the lake shore to 213.4 meters msl (700 feet) inland.

The site stratigraphy consists of glacial deposits that overlay limestone bedrock. The glacial deposits range from sandy gravel and cobble to clay till. A cross-section of the IA's subsurface geology is in Figure 2-5 of the LTP.

The results of the drilling and well installation confirm that fill materials and recent unconsolidated deposits underlie the IA (cover materials) having a total thickness ranging from 12 to 24 meters (40 to 80 feet). Approximately, three to six feet of fill material was added to the land surface in the IA during the plant construction. Fractured limestone bedrock underlies the cover materials. While the surface elevation in the IA remains relatively constant at 179.8 to 180.7 meters (590 to 593 feet) msl, the variation in cover materials' thickness results from a pronounced north to south decrease in the bedrock elevations at the site.

The unconsolidated deposits consist of several units that are laterally continuous across the site. These units comprise a thin upper zone composed chiefly of granular sand and gravel (Units 6a and 6b, 2.4 to 3.7 meters [8 to 12 feet] thick), and a much thicker lower zone composed of a series of units containing a higher proportion of clay and silt (Units 2, 3, and 5; 10 to 25 meters [32 to 76 feet] thick). In addition, an intermediate depth sand layer (Unit 4; 0.6 to 1.5 meters [2 to 5 feet] thick) is also present in this lower zone at depths of 6 to 7.5 meters (20 to 25 feet). The licensee reported a similar sequence of units, including the intermediate depth sand layer, in an earlier study (Ref. 12) conducted approximately 305 meters (1,000 feet) west of the main power station area. This indicates that the unconsolidated deposits found in the main power station area are continuous over a wide area.

The bedrock material (Unit 1) across the site is gray to brown fossiliferous limestone with abundant natural fractures. The bedrock is the Gravel Point Formation, the lowermost member of the Traverse Group of Mid-Devonian age (Ref. 13). Because the pronounced decrease in bedrock elevations from north to south across the site is greater than the regional bedrock dip (40 feet per mile to the south-southeast), the licensee infers that the south bedrock strata were from a slightly lower stratigraphic zone than those encountered on the north side of the site.

Ground Water Regime and Hydrogeology.

CE identified a total of three ground-water-bearing units at the site. These include the shallow ground water unit, the lower portion of the near-surface sand and gravel layer (Unit 6a and 6b); the intermediate depth sand layer unit (Unit 4), and the underlying fractured limestone bedrock unit (Unit 1). The shallow ground water unit is unconfined, while the other two units are confined. The inferred entry point for ground water in the shallow unit is predominantly south of the main power station area in the undeveloped portion of the BRP site. Much of this area is low, wetland type terrain and a significant portion of the precipitation falling in this area likely infiltrates into the soil and enters the shallow ground water unit. The ground water entry points for the other two units are also located to the south, at varying distances from the main power station area. The ground water flow in all three of these units is northerly into Lake Michigan. Figure 2.5 of LTP delineates the relationship between these units.

The fractured limestone bedrock unit consists of separate upper and lower bedrock zones. The lower bedrock zone encountered on the south side of the site (monitoring well PZ-1D) is characterized by primarily horizontal ground water flow from the source area northward toward Lake Michigan. Due to the irregular bedrock topography, the upper bedrock zone encountered on the north side of the site (monitoring wells PZ-2D and PZ-3D) is apparently isolated from the ground water flow from the source area, located to the south, by low permeability unconsolidated sediments and is assumed to receive ground water vertically from the lower bedrock zone.

The near-surface location of the shallow ground water unit (less than 25 feet) would preclude its use as drinking water. The State of Michigan and Charlevoix County health department regulations require a minimum depth of 25 feet for potable ground water supplies; exceptions to this rule are where a deeper ground water source is not available, and where a confining layer is present above the ground water-bearing unit, neither of which apply to this unit.

The intermediate depth sand layer unit (though it is a confined water-bearing unit) does not qualify as potable drinking water source based upon the above regulations (i.e., its depth from the land surface is less than 25 feet and the existence of the deeper potable fractured bedrock unit). Also, the limited thickness of this intermediate unit limits its capacity to provide an adequate supply of drinking water for a homestead.

The fractured limestone bedrock unit is considered the potable drinking water at this site. This unit provides drinking water for private and public users at and near the BRP site.

Radiological Spills, Leaks, and Releases

The licensee, in its LTP and HSA, acknowledged that spills, leaks, and releases of plant-generated radionuclides have occurred at the BRP facility. The licensee responded to the radiological contamination of the soils and ground water with remediation activities and additional characterization. During the evaluation of the effects of radiological contamination on the ground water, the licensee installed monitoring wells in 1994 to evaluate the impact of a major tritium leak observed within the foundation of the Turbine Building. Subsurface leaks in pipes caused the tritium leakage in this area connected to the Condensate Storage Tank in 1984.

The licensee evaluated other spills, leaks, and releases of plant-generated radionuclides within the IA with the 1994 monitoring wells and by installing additional monitoring wells in 2002 and 2004 along the potential ground water pathways of these radionuclides. More details on these spills, leaks, and releases can be found in LTP Section 2.2.5.

Licensee's Response to the Ground water RAIs

The NRC staff generated a request for additional information (RAI) to clarify ground water issues at this site based upon the licensee's previous LTP. The licensee responded to the RAIs by discussing existing hydrogeologic studies of the site and by performing additional characterization of the ground water, emphasizing potential plant-generated radionuclide contamination. The licensee's additional characterization included the installation of monitoring wells screened in Units 3 and 4.

The licensee's revised LTP adequately addressed the NRC staff's ground water concerns. The following material presents the licensee's radiological characterization of the ground water at this site. Additional staff comments are presented to clarify and supplement the characterization.

Radiological Monitoring Wells

During decommissioning planning in 1994, the licensee installed nine shallow ground water-monitoring wells (MW-1 through MW-9). It installed these wells at depths of 6 to 6.7 meters (20 to 25 feet) below the land surface in the IA. The utility of some of these monitoring wells [MW-1 (Units 4 and 6b), MW-5 (Units 4 and 6b), MW-7 (Units 3 and 4), and MW-8 (Units 4 and 6b)] is limited because, according to the Hydrogeological Assessment Report by Otwell Mawby (Ref. 13), they are screened in multiple units). Geologic logs of these monitoring wells are reprinted in Appendix B of Mawby.

In 2002 and 2004, the licensee addressed shortcomings in its radiological monitoring program by installing additional monitoring wells that are screened in only one unit: either one of the three ground water-bearing units or one of the three non ground water-bearing units as previously discussed and delineated in Figure 2-5 of the LTP. In 2002, the licensee's contractor (Mawby) installed 13 monitoring wells that included: monitoring wells PZ-1S, PZ-2S, PZ-3S, PZ-4S, PZ-5S, and PZ-6S in the unconsolidated shallow ground water unit; PZ-1M, PZ-2M, and PZ-3M in the intermediate depth sand layer unit; PZ-1D, PZ-2D, and PZ-3D in the fractured limestone bedrock unit and one additional piezometer (PZ-3Ma) in Unit 3, the aquitard between the intermediate and bedrock units. Geologic logs of these 13 monitoring wells are found in Appendix A of Mawby. In 2004, the licensee installed six additional monitoring wells (PZ-7M,

PZ-7Ma, PZ-8M, PZ-8Ma, PZ-9M, and PZ-9Ma) in Units 3 and 4. It installed these monitoring wells in 2004 to confirm that ground water flow and radionuclide transport (tritium) did not occur west, east, and south of the Turbine Building and the Containment Building.

Potentiometric Ground Water Surfaces and Ground water Flow Directions

The ground water flow patterns within the BRP facility are based upon the potentiometric ground water surfaces, the hydraulic gradients, and hydraulic conductivities of the different rock types. The licensee developed potentiometric maps for the shallow ground water unit and the intermediate depth sand layer unit (several figures in Appendix 2-F, pages 2F-1 through 2F-8, in the LTP) from the May 15, 2002, July 24, 2002, November 4, 2002, and March 13, 2003 sampling events. The licensee did not develop a potentiometric map for the bedrock unit because the difference in completion depths (screened intervals) for the three bedrock wells did not provide sufficient data to do so. The approximate ground water flow directions are toward the northeast (N10E) in the shallow ground water unit and toward the northwest (N19W) in the intermediate depth sand layer unit. The hydraulic gradients based upon the May 15, 2002 sampling event are 0.015, 0.028, and 0.019 for the shallow, intermediate, and bedrock units, respectively.

Investigation of hydraulic head differences evaluated vertical ground water movement using nested monitoring well pairs installed at three locations across the site (PZ-1, PZ-2, and PZ-3). Comparison of the hydraulic heads in the nested monitor wells indicates that vertical head differences occur at all three locations. At the locations south of the Containment Building (PZ-1 and PZ-2), the head differences from nested wells indicate upward vertical components of flow above the intermediate depth sand layer unit and downward vertical flow components below the intermediate depth sand layer unit. At the location north of the Containment Building (PZ-3), the head differences from nested wells indicate downward vertical flow components both above and below the intermediate depth sand layer unit. Vertical hydraulic gradient data vary in magnitude both above and below the intermediate sand layer unit. The overall downward vertical gradient in the area north of the containment sphere indicates the potential for downward migration of mobile radionuclides from the shallow ground water or intermediate depth sand layer units to the fractured limestone bedrock unit.

Infiltration and Recharge

Infiltration of precipitation through the cover materials and recharge to the shallow ground water unit is extremely variable in the IA. Changes in the cover materials and their vertical permeabilities cause this variability and in preferential flow. The fill materials will have the largest infiltration and recharge rates except in areas where extensive preferential flow occurs. The drainage systems and impermeable surfaces from buildings and paving that were added to handle plant activities further complicate the infiltration and recharge within the IA. Infiltration within the IA should be the lowest of any area at this site because of the extensive amount of impermeable surfaces in the IA. However, preferential ground water flow may be significant in this area because extensive infiltration can sometimes occur where the impermeable surface is cracked allowing precipitation and runoff to enter through the cracks.

As previously mentioned, ground water recharge of the shallow ground water unit occurs primarily outside the IA, to the south, east, and west. Ground water recharge of the

intermediate depth sand layer and the fractured limestone bedrock units occurs far to the south of the main power station area, beyond the BRP site boundary.

Ground water and Lake Michigan Fluctuations

Ground water fluctuations for all three ground water-bearing units exhibit a slight seasonal variation, based upon the 2002 and 2003 monitoring data. The seasonal variations result in increased hydraulic heads in the spring or early summer and decreased heads in the late fall or winter. CE estimates the magnitude of seasonal change at 0.12 meters (0.4 feet) for the shallow ground water unit, and at 0.3 meters (1.0 feet) for the intermediate depth sand layer and fractured limestone bedrock units. The potentiometric maps delineated by figures in Appendix 2-F, pages 2F1 through 2F8, in the LTP, illustrate the recent seasonal variations in the hydraulic heads for the shallow ground water and intermediate units. An evaluation of the hydraulic heads over extended time periods indicates that ground water levels have generally decreased from year to year during the past 15 to 20 years.

Lake Michigan water levels also display seasonal and long term water level changes. The annual variations in Lake Michigan water levels are approximately 0.3 meters (1.0 feet) with the maximum occurring June - July and minimum occurring in December - January (Ref. 14). Currently, Lake Michigan is in a low water level episode. Recent data for the past 75 years indicates that lake levels have an approximate 30-year periodicity. Water level changes from tidal effects on Lake Michigan is considered negligible compared to other factors (e.g., weather).

Aquifer Parameters

The aquifer parameters for the BRP facility are based upon slug tests and laboratory analyses performed on the monitoring wells installed in 2002. The licensee performed the slug tests on the 13 monitoring wells installed in 2002 and monitoring well MW-6 to generated hydraulic conductivity values for the screened interval of each well. The laboratory analyses determined the quantitative grain-size distribution, bulk density, total porosity, and effective porosity for selected samples. They also performed laboratory permeability (i.e., hydraulic conductivity) tests on samples from units where reliable slug tests could not be performed.

They conducted the slug tests by removing a known volume of water from each well and measuring the rise in water levels over time. Results of each slug test and a summary table are listed in Appendix C of Mawby (Ref. 13). The mean hydraulic conductivities for the three ground water-bearing units are listed in Table 2 of Mawby. The mean hydraulic conductivity values are 1.76E-03, 3.32E-03, and 4.83E-03 cm/sec (5.0, 9.4, and 13.7 feet/day) for the shallow ground water, intermediate, and bedrock units, respectively.

Laboratory grain-size (sieve) analyses were performed on samples from monitoring wells PZ-4S, PZ-5S, PZ-6S, PZ-1M, PZ-2M, PZ-1D, and PZ-3D; borings HS-1 and TB-1; and two fill samples. Results of the laboratory analyses are listed in Appendix D of Mawby. The engineering properties determined from the laboratory analyses, as well as the slug tests, were as expected except for Units 3 and 5. These units are considered aquitards, but had higher portions of sand (60 to 70 percent) and larger hydraulic conductivities than expected (only 2 orders of magnitude lower than the water-bearing units). These data suggest that the aquitard

layers are only partially confining. Depending upon the hydraulic heads, leakage may occur through these units.

Ground Water Sampling and Analysis for Radionuclides

The licensee analyzed ground water samples from the monitoring wells installed in 1994 and 2002 for tritium, gamma emitting radionuclides, and other hard to detect (HTD) radionuclides that could have potentially originated from plant activities (Table 2-11 of the LTP). The sample analyses identified tritium as the only radionuclide present in the ground water above the minimum detection activity (MDA).

Table 2-12a of the LTP lists additional analyses for tritium for the 1994 through 2004 time period for wells installed in 1994; Table 2-12b lists results for the 2002 through 2004 time period for wells installed in 2002. The tritium results for the three ground water-bearing units indicate that tritium concentrations are largest in the shallow ground water unit and that the tritium concentrations decrease in the deeper ground water-bearing units. Also, the tritium concentrations have decreased over time in the shallow ground water unit; the period of record for the deeper ground water units is not long enough to determine a tritium trend.

The licensee also sampled monitoring wells beneath the Containment Building to determine the movement of radionuclides released during the 1984 condensate system leak beneath the nearby Turbine Building. Ground water samples from these monitoring wells (SC-1, SC-2, SC-3, and SC-5), which they have also listed as Sphere 1, Sphere 2, Sphere 3, and Sphere 5 in the LTP, were analyzed for tritium, gamma spectroscopy, and HTD. The analytical results indicate (Tables 2-11 and 2-12c in the LTP) that they have observed only tritium at concentrations above MDA. In that time, the tritium concentrations exceeded the U.S. Environmental Protection Agency's maximum concentration limit (MCL) value of 20,000 pCi/L by a factor of one to two, but have a generally declining trend.

Tritium concentrations for the BRP site monitoring wells sampled during July 2004 (Ref. 15) are listed in Table 2.1-1. This table contains tritium values for monitoring wells installed in 1994, 2002, and 2004. Results indicate that the tritium concentrations for the monitoring well installed in 1994 and 2002 have not changed significantly over the last few years (Tables 2-12a and 2-12b of the LTP). Also, the wells installed in 2004 (PZ-7M, PZ-7Ma, PZ-8M, PZ-8Ma, PZ-9M, and PZ-9Ma) have tritium concentrations either below 157 pCi/L or only slightly above this value.

The NRC staff collected six split ground water samples from monitoring wells MW-6, MW-9, PZ-3Ma, PZ-3Mb, PZ-3D, and PZ-5S on December 2, 2003. The Oak Ridge Institute for Science and Education (ORISE), the NRC's independent laboratory, performed the following analytical analyses on these samples (Ref. 16): gross alpha, gross beta, gamma spectrometry (Mn-54, Co-60, Zn-65, Ag-110m, I-129, Cs-134, Cs-137, Eu-152, Eu-154, and Eu-155), H-3, C-14, Sr-90, Pu-241, Ni-63, Fe-55, and alpha spectroscopy (Am-241, Cm-243/244, Pu-238, and Pu-239/240). All of these radionuclide analyses were below the minimum detection concentration (MDC) except tritium. The tritium concentrations for ground water samples from wells PZ-3Ma (2,900 pCi/L), MW-6 (2,050 pCi/L), PZ-3D (770 pCi/L), and PZ-5S (310 pCi/L) were above either the MDC or the uncertainties represented by the 95 per cent confidence level. None of these tritium concentrations exceeds EPA MCL of 20,000 pCi/L.

A comparison of the tritium values for the six split ground water samples (i.e., the NRC and BRP ground water samples for the above wells) indicates a close correspondence. The licensee's tritium concentrations were slightly larger (Tables 2-12a and 2-12b in the LTP). The NRC staff concludes that BRP's radiological analytical program is adequately measuring the occurrence of plant-generated radionuclides in the ground water.

Ground Water Movement and Radionuclide (Tritium) Transport

Travel time of plant-generated radionuclide dissolved in the ground water at this facility is limited to tritium movement because tritium has been the only radionuclide observed in ground water above MDC levels.

Tritium is not absorbed by the soils or rock materials; therefore, it acts as a tracer for ground water movement within the BRP facility. The average velocity for ground water can be calculated using the formula:

$$v = Ki / \theta.$$

Where v = average velocity (units - length per time),
 K = hydraulic conductivity (units - length per time),
 i = change in hydraulic gradient or head (units - length per length), and
 θ = effective porosity of the flow medium (no units).

Mawby (2002), in Table 2, lists the average (horizontal) velocities for the shallow, intermediate, and bedrock units as 1.01E-03, 3.20E-03, and 3.09E-03 cm/sec (2.87, 9.08, and 8.76 ft/day), respectively.

The NRC staff conservatively estimated travel times for unobstructed flow after the 1984 condensate system leak from the Turbine Building area to Lake Michigan, approximately 525 feet, for tritium transport in the ground water-bearing units as the following:

1. Shallow ground water unit - 6 months,
2. Intermediate depth sand layer unit - slightly less than 2 months, and
3. Fractured limestone bedrock unit - 5 years.

The actual travel times, however, should be larger because the tritium needs to seep out of the Turbine Building foundation and in some cases move around some of the building foundations.

The licensee in the LTP estimates the travel times of tritium in the shallow ground water unit as 330 days for unobstructed flow to the north and as three years for obstructed flow, first westward around the foundation and then north to Lake Michigan. The downward vertical gradient impacts both ground water paths. However, the horizontal gradient for both paths is greater than the vertical gradient; therefore, the tritium should be discharged into Lake Michigan before it reaches the bedrock unit.

In the LTP, the licensee discusses the tritium flow in Unit 5, the upper aquitard unit, as extremely low about 2 meters/year (6.5 feet/year). The NRC staff agrees with the licensee that

Table 2.1-1. Tritium Concentrations for Monitoring Wells Sampled During July 2004

Monitoring Well	Collection Date	H-3 (pCi/L)
MW-1	07-15-2004	280 ± 95 [†]
MW-5	07-15-2004	5,215 ± 215
MW-6	07-15-2004	2,730 ± 165
MW-7	07-15-2004	< 157 [‡]
MW-9	07-15-2004	367 ± 96
PZ-1D	07-15-2004	<157
PZ-2D	07-20-2004	172 ± 90
PZ-3D	07-21-2004	403 ± 97
PZ-1M	07-15-2004	< 157
PZ-1M (duplicate)	07-15-2004	< 157
PZ-2M	07-20-2004	< 157
PZ-3Ma	07-20-2004	2,671 ± 164
PZ-3Ma (duplicate)	07-20-2004	2,938 ± 170
PZ-3Mb	07-20-2004	724 ± 109
PZ-7M	07-20-2004	< 157
PZ-7Ma	07-20-2004	< 157
PZ-8M	07-20-2004	< 157
PZ-8Ma	07-20-2004	< 157
PZ-9M	07-20-2004	408 ± 97
PZ-9Ma	07-21-2004	199 ± 91
PZ-3S	07-20-2004	203 ± 91
PZ-5S	07-20-2004	2,656 ± 163

† Probable counting error at 95 per cent confidence level.

‡ The less than (<) value is based upon 4.66 counting error for background samples.

his unit would not be considered a significant contributor with respect to potential radionuclide transport.

In the LTP, the licensee discusses another ground water pathway in the intermediate depth sand layer unit (Unit 4) and the lower aquitard unit (Unit 3). Tritium contaminated these units during the 1984 condensate system leak when tritium flowed from the upper units into these units along the Containment Building foundation. The licensee estimates the travel time to Lake Michigan in the intermediate unit as 200 days. The licensee did not discuss the travel time for ground water transport in Unit 3; however, it should be similar to the upper aquitard unit. Therefore, Unit 3 should not be a significant contributor with respect to potential radionuclide transport.

The licensee did not discuss travel time for the bedrock unit. However, it does discuss the downward (vertical) gradient of ground water from the lower aquitard to the bedrock unit as approximately 7.5 meter/year (25 feet/year) with aquitard thickness of 7.5 meters (25 feet).

The licensee provides tritium plumes for the shallow ground water, intermediate, and bedrock units in Figures 2-13, 2-14, and 2-15 of the LTP, respectively. The plume boundary for each unit coincides with the location of 1,000 pCi/L tritium concentration isopleth.

The licensee states in the LTP that remediation of the areas within and near the Containment and Turbine Buildings will remove the tritium source term in the ground water and rock materials. Therefore, the NRC staff concludes that the tritium concentrations in the three ground water-bearing units should decrease significantly over the next few years, considering the half life of tritium and horizontal and vertical dispersion.

Based upon the travel time estimates, tritium has already reached Lake Michigan in each ground water-bearing unit. Further from the tritium source term and closer to Lake Michigan, the tritium concentrations should decrease because of horizontal and vertical dispersion. Currently, the maximum observed concentrations of the tritium plumes are approximately 10 per cent of the tritium MCL in the shallow ground water unit. As the tritium plumes reach Lake Michigan, the much larger volume of water dilutes them. Therefore, the potential dose to humans and the effects on the environment of the tritium plumes reaching Lake Michigan are insignificant.

Ground Water Monitoring During Decommissioning

The licensee agreed to maintain a ground water monitoring program during the decommissioning of the BRP site. This includes quarterly sampling of tritium in the monitoring wells installed in 1994, 2002, and 2004. The licensee also agreed to install replacement wells for those monitoring wells that they abandon during decommissioning because of their location either within or near the decommissioning areas.

After all decommissioning in the IA and the Final Status Survey activities have been completed, the licensee can discontinue the ground water monitoring program if they observe no significant concentrations of radionuclides during the preceding activities.

Conclusion

The NRC staff concludes that the licensee's ground water characterization at the BRP facility with respect to plant-generated radionuclide is acceptable.

The remaining remediation activities at this site are not likely to increase the level of plant-generated radionuclides dissolved in the ground water. Therefore, the potential dose to humans from plant-generated radionuclides at this site is insignificant.

2.1.1.6 Surface Water

The predominant surface water feature at the BRP site is Lake Michigan, which bounds the northern portion of the site. Tritium, a plant-generated radionuclide, dissolved in the ground water has reached Lake Michigan. Based upon the characterization of this site, tritium will continue to reach Lake Michigan; however, the tritium concentrations should continue to decrease over time.

The impact of tritium on the Lake Michigan for past, current, and future times is insignificant. Once tritium reaches Lake Michigan, it is diluted to concentrations that should not create any environmental problems.

Also, the likelihood that other plant-generated radionuclides will reach Lake Michigan is extremely low.

The NRC staff concludes that the licensee's surface water characterization is acceptable.

2.1.1.7 Sediment

The presence of radioactive contamination in sediment was assessed using the process designed for soils. The licensee included portions of the site with sediments in characterization surveys. These areas include shorelines west of the discharge canal, and storm water drains. The licensee analyzed sediment samples for the presence of radioactivity and radionuclide distributions, using the same characterization process described for soils.

Sediment sampling and analysis revealed varying contamination levels, depending on whether plant operations, including spills and leaks had affected portions of these areas. The contaminants were mainly Co-60 and Cs-137. CE reported contamination levels to be low, ranging from non-detectable levels to about 0.67 Bq/g (17.7 pCi/g) for Co-60, and from non-detectable to 122 mBq/g (3.3 pCi/g) for Cs-137.

The staff concludes that the approach used by the licensee to characterize sediments is acceptable.

2.1.1.8 Pavement

As with sediments, the presence of radioactive contamination under paved areas was assessed using the process designed for soils. Characterization surveys included portions of the site where CE expected pavement and underlying soils to be contaminated. Paved roadways were determined to be Class 3 areas. The licensee will remove all other pavement, including the parking lots and asphalt within the protected area for disposal. The LTP identified that some

paved areas and covered soils are suspected of being contaminated. However, these areas are yet to be fully characterized and evaluated by the licensee.

The staff concludes that the approach used by the licensee for the initial characterization of contamination levels of paved areas is acceptable, while recognizing that further efforts will be necessary to complete the characterization. This information will be used to identify areas that might require remediation, as described under continuing characterization activities. The staff will confirm, during in-process inspections and confirmatory surveys, that CE appropriately evaluated residual contamination levels in paved areas and those areas meet the release criteria for surface or deep soils.

2.1.2 Site Characterization - Summary Finding

The staff reviewed the information in the LTP for the BRP Nuclear Plant, according to Section B.2 of NUREG-1700 (Ref. 17). Based on this review, the staff concludes that the licensee met the objectives of providing an adequate site characterization as required by 10 CFR 50.82(a)(9)(ii)(A).

2.2 Remaining Site Dismantlement Activities

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(B), the licensee provided the status of dismantlement and decontamination activities for the BRP major systems, structures, and components, as of March 2003. Also, in accordance with the guidance provided in NUREG-1700, and Regulatory Guide 1.179 (Ref. 18), the licensee provided the following: a radioactive waste characterization; an estimate of the quantity of radioactive material that it will ship for offsite burial; an estimate of personnel exposures; and the methods that it will use to control the spread of contamination while performing dismantlement activities. In addition, the licensee provided descriptions of the remaining dismantlement activities that were sufficiently detailed to allow the staff to plan inspections during dismantlement.

There are three phases of decommissioning activities: (1) decontamination and demolition of structures; (2) soil remediation; and (3) final status surveys. CE describes the first phase in Chapter 3 of the BRP LTP. Many of the major decommissioning activities have already been completed. Chapter 3, Table 3-2, identifies each completed activity and lists the completion date; Table 3-5 lists the remaining activities and projected completion dates. Chapter 4 describes soil and water remediation, and Chapter 5 describes the final status surveys.

The total volume of waste projected for BRP decommissioning is 17,333 cubic meters (612,100 cubic feet), which includes 2,042 cubic meters (72,100 cubic feet) of radioactive waste and 15,291 cubic meters (540,000 cubic feet) of demolition debris with no detectable radioactivity. The licensee also estimated the total worker exposure during decommissioning to be 7.0 person-sievert (700 person-rem), which is less than the 18.7 person-sievert (1874 person-rem) found acceptable in NUREG-0586, Supplement 1, Table 4-1(Ref. 19). The licensee committed to continue the existing BRP Radiation Protection Program and the Radioactive Waste Management Program; this includes the existing program used to control the spread of contamination while conducting dismantlement activities.

The staff reviewed the LTP against the information in Section B.3 of NUREG-1700. Based on this review, the staff concludes that the licensee identified, in sufficient detail, the remaining dismantlement activities necessary to complete decommissioning of the facility, as required by 10 CFR 50.82(a)(9)(ii)(B) and 10 CFR 50.82(a)(11)(i). Further, the staff concludes that the remaining dismantlement activities can be completed in accordance with 10 CFR 50.59.

2.3 Plans for Site Remediation

In accordance with the requirement of 10 CFR 50.82(a)(9)(ii)(C), the licensee provided its plans for completing radiological remediation of the site. The licensee plans to remediate the site, including structures and systems that remain on the site, to the criteria of 0.25 mSv/yr (25 mrem/yr) for all pathways, which is the unrestricted use criterion specified in 10 CFR Part 20, Subpart E. To meet this criterion, the licensee is using typical remediation methods, including chemical decontamination, wiping, washing, vacuuming, scabbling, chipping, and abrasive blasting. The licensee will send all structures and systems to an offsite processing facility, or to a low-level radioactive waste facility for disposal.

As specified in Part 20, Subpart E, a site is acceptable for unrestricted use if the remaining residual radioactivity results in a TEDE less than or equal to 0.25 mSv (25 mrem) per year above background, and the remaining residual radioactivity is reduced to levels that are ALARA. The licensee provided its ALARA analysis process. CE describes the ALARA analysis in Chapter 4 of the BRP LTP. The licensee's formulas for calculating the remediation levels conform to the guidance provided in NUREG-1727 (Ref. 20).

The staff compared the information in the LTP against Section B.4 of NUREG-1700 and against similar decommissioning activities conducted at other plants. Based on this review, the staff concludes that the licensee met the requirements of 10 CFR 50.82(a)(9)(ii)(C) by providing a detailed discussion of the remediation plans for the remaining decommissioning activities.

2.4 Final Site Survey

Licensees perform the FSSs after they have characterized an area, remediation has been completed, and the licensee believes that the area is ready to be released for unrestricted use. The purpose of the FSS is to demonstrate that each area, as defined by survey classifications, meets the radiological criteria for license termination. The FSS design entails an iterative process that requires appropriate site classification - based on the potential residual radionuclide concentration levels compared with the DCGLs - and formal planning using data quality objectives (DQOs). An integrated design addresses selection of appropriate survey and laboratory instrumentation, well-defined survey methods and procedures, and statistically based measurement and sampling plans for collecting and evaluating the FSS data. NRC guidance for planning, designing, implementing, and evaluating FSSs is presented in four key documents¹: (1) MARSSIM; (2) NUREG-1727, Appendix E - "Implementing the MARSSIM

¹ In September 2003, The NRC issued its "Consolidated NMSS Decommissioning Guidance," NUREG-1757, in three volumes. This document incorporates all previous guidance on decommissioning for materials sites, and on dose analysis for all sites, including reactors.

Approach for Conducting Final Radiological Survey” (NMSS SRP); (3) NUREG-1505, Rev. 1 (Ref. 21); and (4) NUREG-1507).

The LTP presents the framework through which CE will plan, design, and implement all FSSs. The NRC evaluated the following relevant sections of the LTP: Section 5.2 - “Development of Survey Plan”; Section 5.3 - “Survey Design and Data Quality Objectives”; Section 5.4 - “Survey Methods and Instrumentation”; Section 5.5 - “Data Collection and Processing”; Section 5.6 - “Data Assessment and Compliance”; Section 5.7 - “Reporting Format”; and Section 5.8 - “FSS Quality Assurance Plan.”

The NRC will conduct performance-based, in-process inspections throughout the decommissioning activities. The purpose of the inspections is to review the procedures, methodology, equipment, training and qualifications, and QA and QC measures. The NRC conducts in-process inspections approximately monthly. Since submittal of the LTP, The NRC has issued seven inspection reports for the BRP site, the most recent is IR 05000155-04-001. The inspections identified no violations.

The licensee used site characterization data, process knowledge, and operational and routine surveillance survey records, as the principal means for initially classifying site areas as impacted or non-impacted. Table 5-2 and Figures 5-2 and 5-3 identify site areas and structures by survey unit; the approximate size of each survey unit; and the initial classification (as Class 1, 2, or 3). Section 5.2.2 presents the process CE will use to classify site grounds as FSS units. The proposed survey unit sizing and classification process for site grounds are consistent with NRC guidance provided in NUREG-1575, usually referred to as MARSSIM, and NUREG-1727.

The LTP describes information and parameters that CE will apply in developing DQOs, as defined in MARSSIM. The elements of the DQOs include: the null hypothesis (i.e., the survey unit does not meet the release criteria); decision errors; selection of an appropriate statistical test; limits on decision errors; scan coverage as a function of survey unit classification; variables for calculating sample size and sampling density for each survey unit; sampling locations and reference grid system for buildings and grounds; survey design process; and establishing background radiation levels in selected reference areas. Variables used to calculate sample sizes are the DCGL, lower boundary of the gray region (LBGR), and estimates of the variability of the contaminants in a survey unit (commonly referred to as “sigma”). The statistical tests discussed in NUREG-1575 are the Wilcoxon Rank Sum test and the Sign test. The LTP states that the licensee will apply the Sign test, implemented by using the unity rule and a method using surrogate radionuclides. The LTP presents information (Section 2.3.3) that shows Cs-137 background reference area measurements are not needed.

The input parameters for sample size calculations include the DCGL; the LBGR (which generally provides an estimate of the mean concentration in the survey unit, but may be adjusted to optimize the design); and an estimate of the radionuclide variability. These parameters, together with decision errors, are used to calculate the required number of statistical samples. For initial planning purposes, the licensee set the LBGR at 50 percent of the DCGL. CE will develop the estimated variabilities (sigmas) from survey data that utilize identical measurement techniques as the FSS. Default decision errors are set at 0.05 for both Type I and II errors. The principal decision error of concern to NRC, for the stated null

hypothesis, is the Type I or α error. This error occurs when a survey unit is determined to meet the release criteria when in fact it does not. The default value of 0.05 for the Type I or α error used by BRP is acceptable. In the LTP, CE commits to determine sampling density using MARSSIM guidance. In addition, they will adjust the number of samples to reflect differences between instrumentation scan minimum detectable concentrations and DCGLs. The NRC concludes that the approach and statistical survey planning discussed in the LTP are acceptable.

LTP Section 5.3.5 and Figure 5-4 in the LTP present the site coordinate system and survey unit grid that CE will use for keying all areas during the conduct of FSSs. The NRC concludes that the proposed approach in coupling survey units with the site grid, using site and global positioning system (GPS) coordinates to record specific locations, is appropriate for the survey unit classification and the type of survey unit. Grid coordinates will then serve as the basis for the random-start systematic sample-location selection.

CE discusses selection of survey instrumentation (rate meters and detectors), calibration, and survey methods in Section 5.4.3 of the LTP. The selection process will ensure that the instrumentation CE uses for the FSSs will respond adequately to the types of radiation being emitted by the various radionuclides of concern. It will also ensure that the instruments are sufficiently sensitive to detect these radionuclides, or gross activity, at levels within appropriate fractions of the DCGLs. And it will ensure that they calibrate instruments in a manner that accounts for the expected or known radionuclide mix, expected radiation emission energies of the mixture, surface efficiencies, and how the contaminants are physically distributed in the media. The NRC concludes that the list of instrumentation and the basis for instrumentation detection efficiencies listed in Table 5- 8, are appropriate for the primary radionuclides identified in Section 2.5.3.

Table 5-8 of the LTP presents typical efficiencies for the primary types of detectors that CE will use in FSSs. Moreover, the licensee indicated the possible use of *in situ* gamma spectrometry in certain circumstances. At this time it does not include details in the LTP for this survey method; the licensee stated that it will provide a technical basis document to the NRC for review and approval once such a need has been identified and the appropriate survey procedure has been developed.

For volumetrically distributed contamination in soils and concrete, the licensee will account for HTD radionuclides that may be present through a modified DCGL, using a surrogate approach. The radionuclides of concern include H-3, Mn-54, Fe-55, Co-60, Sr-90, Cs-137, Eu-152, Eu-154, and Eu-155 (Table 6-10). The process will consider the basis of the dose model, described in Section 6 of the LTP, DCGL for the media, and radionuclide profiles and fractions given in Section 2.4.5.4. In each instance, detector response and MDC will need evaluation to confirm that the survey techniques will be adequately sensitive to measure activity levels at the modified DCGL. NRC review of these sections of the LTP concludes that the proposed approach is acceptable.

CE discusses the conduct of routine operational checks and calibration procedures in LTP Sections 5.4.3.2, 5.4.3.3, and 5.8.2.4. The licensee will use National Institute of Standards and Technology (NIST) traceable calibration sources that are similar in energy to the primary radionuclides of concern. The licensee will perform instrument response checks before use

each. Should a response check fail the +/-10 percent criteria for a portable survey instrument, the licensee will remove it from service, and any data collected since the last acceptable check will be evaluated and may be discarded. For laboratory instrumentation, the licensee will monitor responses using control charts and appropriate radioactive standards. NRC review of these sections of the LTP concludes that the proposed approaches are acceptable.

The method for conducting FSSs is in Section 5.4 of the LTP; it is supplemented with information in Section 5.5. Section 5.4 discusses methods for performing surface scans and direct measurements, and sampling methods and sample size. If contamination below 15 cm is suspected or known, samples will be collected using an auger or equivalent method. If CE has already excavated a survey area and remediated to the soil DCGL, it will treat this area as a surface soil, and perform the FSS on the excavated area. Because all buildings and structures used during nuclear plant operations are scheduled for removal, the majority of the survey units within the Industrial Area will include excavated areas. The licensee may utilize radiation detectors of sufficient sensitivity in a "down hole" configuration to identify the presence or absence of subsurface contamination, and the extent of such contamination. If the detector identifies the presence of contamination at a significant fraction of the DCGL, CE will perform confirmatory laboratory analyses of soil samples of the suspect areas. CE committed to performing all subsurface sampling in accordance with the guidance in Section 11.1 of Appendix E of NUREG-1727. The sample size for subsurface samples will be determined using the same methods described for surface soil. Per NUREG-1727, scanning is not applicable to subsurface areas; however, BRP FSSs will employ scanning techniques commensurate with the survey unit classification. Scanning on subsurface soils, where accessible as an excavated surface (e.g. trench walls), will demonstrate compliance with site release criteria.

The scan coverage for surface soils is based on survey unit classification. Class 1 survey units will receive 100 percent scan coverage; Class 2, 10 to 100 percent; and Class 3 up to 10 percent. They currently perform soil and bulk-material samples at locations defined using methodology from MARSSIM. The licensee stated that measurement and sampling locations are determined on a random-start systematic pattern for Class 1 and Class 2 survey units, and randomly for Class 3 survey units; all will be based on the reference grid. Additional measurements or samples may also be collected from areas of elevated radioactivity detected during scanning. In Class 2 and Class 3 survey units, scan coverage will vary depending on conditions and a review of the data used in developing survey design specifications. The licensee may supplement scanning and measurement activities with other measurement techniques and sampling. These proposed methodologies for surveys are acceptable and generally follow NRC guidance.

Section 5.3.6 presents the approach that CE will use to develop investigational levels. It includes the process to investigate areas found to contain elevated levels of activity above the DCGL of the applicable investigational levels. It also identified the actions it will take once they have confirmed that an action level has been exceeded. The process outlines procedural steps regarding the initial detection, investigation process, re-surveying, and comparison of results with the Elevated Measurement Comparison Test; it includes the decision process in determining the need for further remediation, and if the reclassification of the survey unit is warranted. The criteria identified in this process consider: (i) the assumptions made in the survey unit classification; (ii) the most likely or known cause of the contamination; and (iii) the effects of summing multiple land areas with elevated activity within the survey area. Depending

on the results of the investigation, the licensee may reclassify a portion of the survey unit, given sufficient justification. CE established that this evaluation process avoids the unwarranted reclassification of an entire survey unit, while still requiring an assessment as to extent and reasons for the elevated area. If an individual survey measurement (scan or direct) in a Class 2 survey unit exceeds the DCGL, the survey unit or a portion of it may be reclassified; the survey will be redesigned and executed accordingly. A Class 2 buffer zone will typically bound the new Class 1 area to provide further assurance that the reclassified area completely bounds the elevated area. If an individual soil sample in a Class 3 survey unit exceeds 0.5 DCGLw, CE will evaluate the survey unit, or portion of a survey unit, and if necessary, reclassify it as Class 2, and redesign and execute the survey accordingly.

LTP Section 5.2.2 stipulates that the licensee may reclassify survey units to a more restrictive classification. The licensee will document the reasons for reclassification and evaluate the potential for programmatic deficiencies in the survey unit classification process. The approaches proposed for investigating and reclassifying survey units are acceptable.

Sections 5.5 and 5.6 of the LTP discuss the process for data collection and processing, and data assessment. The process includes controls (i.e., using a chain of custody records), for samples collected for analysis. The process describes methods for data analyses, data verification and validation, graphical review of survey data using posting and frequency plots, and application of the Sign test in demonstrating compliance. They will use graphical data representations to identify spatial patterns and potential anomalies that would indicate additional investigation is required. The last step addresses data assessment, statistical tests, use of the unity rule, interpretation of measurements results, and use of the data to reach specific conclusions based on MARSSIM criteria. The approach proposed for evaluating measurement results, using MARSSIM statistical tests against the appropriate DCGLs, and determining compliance, is acceptable.

Section 5.7 of the LTP provides a brief description of the FSS documentation. Information that CE compiles for each survey unit includes a history file and release record. At project completion, the licensee will prepare an FSS report summarizing survey data results and overall conclusions, as they relate to the radiological criteria for each survey unit or groups of survey units. The planned presentation of the site's final radiological status appears to be acceptable and generally consistent with guidance given in the NMSS Standard Review Plan (SRP). However, the adequacy of the site documentation process cannot be determined until the licensee has an opportunity to begin compiling FSS records and assembling FSS reports.

Section 5.8 of the LTP present project management controls and outlines of administrative procedures that CE will use to process areas of the plants from remediation activities to a stage where it can conduct FSSs. CE structures the management along defined project functions, organizations, and responsibilities, by assigned functions. CE discusses the major job functions in Section 5.8.2.1

Section 5.2.4 presents a process that the licensee will use for transferring survey units from the remediation activities to a phase in which it can plan and conduct FSSs without interferences that might lead to questionable survey measurements. The process includes considerations for the isolation and control of survey units to prevent re-contamination. The licensee couples the overall process with QA and QC measures, based on 10 CFR Part 50. LTP Sections 5.8.2.3 to

5.8.2.6 identify specific criteria, including training and qualification, quality control surveys, instrument calibration, data management, and corrective actions. NRC concludes the approach the licensee proposed is appropriate.

The staff reviewed the information provided in the BRP LTP according to Section B.5 of NUREG-1700. Based on this review the staff concludes that the licensee conformed to the requirements of 10 CFR 50.82(a)(9)(ii)(D). The final radiation survey plan proposed in the LTP provides assurance that residual radioactive contamination levels will meet the criteria specified in Part 20, for unrestricted use. However, the staff will confirm, during in-process inspections and confirmatory surveys, that the licensee plans and implements FSSs in accordance with approved plant procedures and that residual contamination levels meet the release criteria.

2.5 Compliance with Radiological Criteria for License Termination

Chapter 6 of the LTP describes the development of residual radionuclide concentration levels that the licensee will use to demonstrate compliance with the regulations for releasing the BRP Restoration Project site. CE selects unrestricted-release screening criteria for the majority of the site; the HSA and the site characterization have indicated there is no residual radioactivity distinguishable from background radiation. Furthermore, CE considers a modified residential farming scenario to develop site-specific DCGLs for the BRP Restoration Project Industrial Area; this is the portion of the site for which residual radioactivity above background is identified in surface and subsurface soil and ground water. CE selects to use the RESRAD computer code, Version 6.21, to model doses resulting from exposures to contaminated soils and ground water.

2.5.1 Site Release Criteria

In the LTP, CE proposes to release the site, after removal of all structures except the lake water intake pipe, septic drainfield, and those supporting ISFSI operations, for unrestricted use per 10 CFR 20.1402. In accordance with 10 CFR 20.1402, the site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent (TEDE) to an average member of a critical group that does not exceed 25 mrem/yr (0.25 mSv/yr), including that from ground water sources of drinking water. Furthermore, when calculating TEDE to the average member of the critical group the licensee shall determine the peak annual TEDE expected within the first 1000 years after decommissioning. 10 CFR 20.1402 also requires licensees to reduce residual radioactivity to levels that are ALARA in order for the site to be considered acceptable for unrestricted use.

The critical group is defined in 10 CFR 20.1003 as “. . . the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for an applicable set of circumstances.” The concept of the average member of the critical group is an attempt to account for the uncertainty and assumptions needed to calculate potential future doses, while limiting boundless speculation on possible future exposure scenarios. Furthermore, the use of the average member of the critical group acknowledges that any hypothetical “individual” used in the dose assessment is based, in some manner, on the statistical results from data gathered from groups of individuals.

2.5.2 DCGLs

One acceptable approach for the licensee to provide reasonable assurance that the final residual radionuclide concentrations will meet the dose criterion specified at 10 CFR 20.1402 for the average member of the critical group is to derive, and commit to, nuclide-specific concentration limits (i.e., DCGLs) equivalent to the dose limit. The DCGL derivation can use either generic screening criteria or site-specific analyses. CE documents the manner in which they derive DCGLs for BRP in Chapter 6 of the LTP. The licensee may elect to demonstrate compliance with the dose criterion for the majority of the site by comparison of the FSS results with published unrestricted release screening criteria; these are specified in NUREG-1727, "NMSS Decommissioning Standard Review Plan," SRP 5.1 (NRC, 2000). Furthermore, where they have identified residual radioactivity in surface or subsurface soil and ground water within the Industrial Area of the site, the licensee developed site-specific DCGLs. NRC staff, using NUREG-1727, reviewed the demonstration of compliance as part of BRP Restoration Project's LTP. In demonstrating compliance, the licensee relies upon a reasonable combination of conceptual models, exposure scenarios, mathematical models, and input parameters to calculate DCGLs. Furthermore, the licensee adequately considered the uncertainties inherent in the modeling analysis.

The NRC staff concludes that the dose modeling for the preferred approach is reasonable and appropriate for the exposure scenarios under consideration. In addition, the approach provides reasonable assurance that the dose to the average member of the critical group is not expected to exceed the criterion in 10 CFR 20.1402. This conclusion is based on the modeling effort performed by the licensee and independent analyses performed by the staff.

2.5.2.1 Screening DCGLs

The licensee proposes the use of published NRC screening criteria to demonstrate compliance with the unrestricted release criteria for the majority of the site. The licensee reports, in the LTP, that the historical site assessment and site characterization indicate residual radioactivity distinguishable from background levels (in surface soils, subsurface soils, or in subsurface water) is not expected for these portions of the site. The majority of the BRP site includes survey areas that the licensee classified as non-impacted or impacted class 2 or 3, consistent with NUREG-1575 (MARSSIM).

The NRC staff reviewed the approach for use of published NRC screening criteria as part of the review of the BRP LTP using NUREG-1727, SRP 5.1. The staff concludes that the dose estimate using the published screening criteria is appropriate for unrestricted release. In addition, the dose estimate provides reasonable assurance that the licensee will meet the dose criterion specified in 10 CFR 20.1402. This conclusion is based on the modeling effort performed by the staff in initially developing the published screening criteria. In determining the dose to the average member of the critical group, the licensee uses the assumptions inherent in the screening analysis and the parameter uncertainties that the NRC staff previously evaluated on a generic basis as part of establishing the default screening analysis.

2.5.2.2 Site-Specific Industrial Area DCGLs

The licensee developed site-specific DCGLs to demonstrate compliance with the unrestricted release criteria for the Industrial Area of the site. The licensee reports in the LTP that the HSA and site characterization indicate residual radioactivity distinguishable from background is identifiable in surface and subsurface soil and ground water within the Industrial Area of the BRP site. Therefore, these areas do not meet the criteria for the use of unrestricted release screening criteria and the licensee developed DCGLs in accordance with the description and justification presented in Chapter 6 of the LTP.

The NRC staff reviewed the dose modeling analyses for unrestricted release as part of the review of the BRP LTP using NUREG-1727, SRP 5.1. The staff concludes that the dose modeling for unrestricted release is reasonable and is appropriate for the exposure scenario under consideration. In addition, the dose estimate provides reasonable assurance that the dose to the average member of the critical group is not likely to exceed the 0.25 mSv (25 mrem) annual dose criterion in 10 CFR 20.1402. This conclusion is based on the modeling effort performed by the licensee and the independent analyses performed by the NRC staff. In determining the dose, the licensee uses a reasonable combination of the conceptual model, exposure scenario, mathematical model, and input parameters to calculate a reasonable estimate of dose. The licensee adequately considered the uncertainties inherent in the modeling analysis. Details regarding the NRC staff review of the licensee's treatment of specific components (i.e., source term, exposure scenarios, mathematical model, and uncertainty and variability) of the derivation of site-specific DCGLs are provided in subsequent sections of this report.

2.5.3 Source Term

The licensee reports published screening criteria for 24 potential radionuclides of concern, listed below in Table 2.5.1, and site-specific DCGLs for six radionuclides observed in BRP environs and three europium isotopes, listed in Table 2.5.2. They determine potential radionuclides of concern from a screening analysis that considers half-life, activity fractions of radionuclides identified in boiling water reactor (BWR) internals at shutdown, and detection of radionuclides at the BRP site. The licensee considers radionuclides identified in NUREG/CR-4289, "Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants," which investigates residual radionuclides at seven nuclear power plants. The licensee discounts those radionuclides with half lives less than 244 days. All the radionuclides identified in NUREG/CR-4289 and discounted for BRP because of short half-life will have decayed through 10 half-lives since the facility shutdown.

Additionally, the licensee considers radionuclides identified in NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials," that would be present in BWR internals at shutdown. The licensee discounts individual potential radionuclides with an estimated activity less than 0.1 percent of total activity published in NUREG/CR-3474 for BWR internals. Furthermore, the licensee evaluates the expected contribution to dose for the discounted radionuclides based on the activity fractions from NUREG/CR-3474. The evaluation indicates that they expect the discounted radionuclides to contribute less than 1 percent of the estimated dose at the year of maximum dose following site release. This is consistent with guidance contained in Appendix E of NUREG-1727. The NRC staff's assessment confirms the licensee's evaluation of potential

dose contribution. Following these screening analyses, the licensee identifies one additional potential radionuclide of concern, Zn-65, identified in NUREG/CR-3474 and not included in the list developed from NUREG/CR-4289.

The licensee combines the potential radionuclides identified in NUREG/CR-4289 and an additional potential radionuclide identified in NUREG/CR-3474 to form a single list of potential radionuclides. CE discounts radionuclides from the combined list that they have not observed in historical 10 CFR Part 61 waste stream characterization analyses. The licensee reports waste stream characterization analyses have not identified Nb-94, Ru-106, Sb-125, Ce-144, and Np-237 above their minimum detectable activity concentrations. The licensee's analysis indicates these discounted potential radionuclides would contribute less than 1 percent of the estimated dose per year at the year of maximum dose following site release. Section 2.5.5.2.2.1 of this report provides more detail of the NRC staff review of this calculation. The staff finds the licensee's approach to discount radionuclides is consistent with guidance contained in Appendix E of NUREG-1727. The licensee's waste stream characterization analyses have identified Cm-243, which they did not include in the initial combined list of potential radionuclides. Therefore, the licensee includes Cm-243 in the list of potential radionuclides of concern for the BRP site. The licensee may elect to use NRC-published screening criteria for the list of potential radionuclides of concern for the majority of the BRP site; it comprises the non-impacted or impacted Class 2 or 3 survey units.

Table 2.5.1. Radionuclides for which published screening criteria are used for the majority of the BRP Restoration Project site.

H-3	Sr-90	Eu-155
C-14	Tc-99	Pu-238
Mn-54	Ag-110m	Pu-239
Fe-55	I-129	Pu-240
Ni-59	Cs-134	Pu-241
Co-60	Cs-137	Am-241
Ni-63	Eu-152	Cm-243
Zn-65	Eu-154	Cm-244

During the site characterization process, the licensee analyzed soil samples for all of the potential radionuclides of concern except tritium. However, they have identified tritium in samples from the three subsurface ground water aquifers. The site characterization analyses have detected six radionuclides from the list of potential radionuclides of concern. Therefore, for the development of site-specific DCGLs for the Industrial Area, the licensee discounted all radionuclides from the list of potential radionuclides of concern that they did not observe during the site characterization process. In addition to the six radionuclides observed during the site characterization process, the licensee elects to include three europium isotopes in the development of site-specific DCGLs for the Industrial Area; this is because of the potential for soil contamination from activated concrete demolition debris.

Table 2.5.2. Radionuclides for which site-specific DCGLs are developed for the Industrial Area of the BRP Restoration Project site.

H-3	Co-60	Eu-152
Mn-54	Sr-90	Eu-154
Fe-55	Cs-137	Eu-155

2.5.4 Exposure Scenarios

The licensee uses the residential farming scenario to demonstrate compliance with published screening criteria for the majority of the site. The licensee considers exposure pathways consistent with the residential farming scenario including direct exposure from residual radioactivity in soils, internal exposure from inhalation of airborne radionuclides, and internal exposure from ingestion of the following: (1) plant foods grown in the soil with residual radioactivity and irrigated with contaminated water, (2) meat and milk from livestock fed with contaminated fodder and water, (3) drinking water from a contaminated well, (4) fish from a contaminated pond, and (5) soil with residual radioactivity. This scenario and associated pathways are consistent with the generic scenario used for development of the screening criteria published in NUREG-5512, "Residual Radioactive Contamination from Decommissioning" and NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination."

In the development of site-specific DCGLs for contamination in soil and ground water, the licensee considers a modified residential farming scenario. In the licensee's modified residential farming scenario, it considers all the exposure pathways described above for the residential farmer except the farmer does not consume animal products generated onsite (i.e., meat and milk). The licensee contends use of the site for subsistence farming is unlikely given the Industrial Area's Lake Michigan shoreline location based on soil quality and regional land-use and economic trends. Currently, the licensee reports there are no lakeshore farms within 20 miles of BRP because of poor soil quality and high economic value of Lake Michigan shoreline property. Furthermore, the licensee calculates that the estimated peak doses from individual site-specific radionuclides occur within the first few years. This is well before 100 years recommended in NUREG-1727, Appendix C, for modifications resulting from land-use patterns to generic scenarios. Therefore, the selection of the modified residential farming scenario for development of site-specific DCGLs for the Industrial Area appears reasonable and consistent with guidance in NUREG-1727, SRP 5.2.

2.5.5 Mathematical Model

For most of the BRP Restoration Project site, the licensee may select published screening criteria to demonstrate compliance with 10 CFR 20.1402 for 24 potential radionuclides of concern. Table 6-8 in the LTP lists DCGLs that are based on published screening criteria for 24 potential radionuclides of concern at BRP. CE may use these to demonstrate compliance for the majority of the site, which is classified according to NUREG-1575 as either non-impacted or impacted Class 2 or 3 survey units.

For the Industrial Area of the BRP site, the licensee selects RESRAD Version 6.21 to develop site-specific DCGLs for the modified residential farming scenario. The licensee reports the results from RESRAD for each of the six radionuclides observed in BRP environs and the three europium isotopes in units of mrem per pCi/g. They scale the results to meet the TEDE unrestricted release criterion. They reduce this from 25 mrem/yr to account for observed tritium contamination in the three ground water aquifers and the discounted radionuclides, to determine acceptable DCGL values. Table 6-10 in the LTP lists DCGLs that CE will use for residual radioactivity in surface and subsurface soils.

2.5.5.1 Screening DCGL Mathematical Model

Consistent with guidance in NUREG-1727, Appendix C, the use of published screening criteria for sites with surface soil contamination may be considered appropriate where: (1) the initial residual radioactivity after decommissioning is in the top layer of the surface soil, (2) the unsaturated zone and ground water are initially free of contamination, and (3) the vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate. NRC staff confirms, from the information published in the LTP, that the appropriate conditions appear to be satisfied for the survey units the licensee identifies as Non-Impacted or Impacted Class 2 or Class 3 outside of the Industrial Area at the BRP site.

2.5.5.2 Site-Specific Industrial Area DCGL Mathematical Model

As stated previously, the licensee selects RESRAD Version 6.21 to develop site-specific surface and subsurface DCGLs for each of the six radionuclides observed in BRP environs plus the three europium isotopes. The licensee also uses RESRAD Version 6.21 and comparison with 40 CFR 141.16 to determine peak doses from observed tritium contamination in three underlying ground water aquifers. The licensee uses the estimated peak tritium doses from the tritium-contaminated aquifers and discounted radionuclides to reduce the 25 mrem/yr dose limit to 24.18 mrem/yr; they use this number to determine site-specific DCGLs for the Industrial Area. NRC staff reviewed the licensee's modeling approach and finds it to be reasonable and acceptable.

2.5.5.2.1 Site-Specific Industrial Area Input Parameters

The licensee develops site-specific Industrial Area input parameters based on the process described in section 6.7 of the LTP. From this, they choose appropriate values for those input parameters that have significant influence on radiation dose estimates. The selection process is consistent with guidance presented in NUREG/CR-6676, "Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes," NUREG/CR-6692, "Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes: User Guide", and NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes".

The licensee, consistent with NUREG/CR-6697, classifies RESRAD input parameters as one of the following three types: behavioral, metabolic, or physical. Behavioral parameters depend on the behavior of the receptor and the scenario definition. Metabolic parameters represent the metabolic characteristics of the receptor and are independent of the scenario definition. Physical parameters are those that would not change if different receptors are considered.

The licensee elects to use the default deterministic input values with limited justification for behavioral and metabolic input parameters. This approach is consistent with guidance in NUREG-1727, Appendix C, when the values are consistent with the generic definition of the average member of the critical group and the screening group reflects the scenario. The licensee considers a modified residential farming scenario in which the receptor is subject to all the exposure pathways associated with the generic residential farming scenario except those associated with ingestion of contaminated meat and milk products. Therefore, this approach appears consistent with the generic resident farming scenario, and the NRC staff finds this approach reasonable and acceptable.

Where site-specific data is available, the licensee incorporates direct measurements for physical and mixed-physical (parameters classified as physical and behavioral or metabolic) parameters. If direct measurement could not determine a physical parameter value, the licensee derived a site-specific value via probabilistic sensitivity analysis using RESRAD Version 6.21. The licensee ranks the physical parameters for which direct measurements are not available in order of their respective importance in dose modeling according to NUREG/CR-6697, Attachment B. They use three levels of priority (i.e., 1, 2, and 3), where 1 represents high priority, 2 represents medium priority, and 3 represents low priority.

The licensee uses the default value used in RESRAD Version 6.21 for all physical parameters ranked as priority 3 except parameters associated with humidity in air, indoor time fraction, and inhalation rate. The licensee developed statistical parameter distributions for these three priority 3 physical parameters similarly to the development process used for priority 1 and 2 physical parameters, described next.

For physical parameters ranked as priority 1 and 2 and those priority 3 parameters mentioned previously, the licensee assigns a generic distribution obtained from NUREG/CR-6697, Attachment C. Based on the results of the sensitivity analysis, the licensee assigns values to input parameters not previously assigned default values or site-specific direct measurements according to the following criteria:

- For Priority 1 and 2 (including the three previously identified priority 3 parameters) physical parameters that are sensitive (i.e., with absolute partial rank correlation coefficient (PRCC) values greater than 0.25) the licensee assigns conservative values. The licensee selects the 75th quartile for positive correlations and 25th quartile for negative correlations.
- For Priority 1 and 2 (including the three previously identified priority 3 parameters) physical parameters shown to be insensitive (i.e., with absolute PRCC values less than 0.25) they assign the median value from their generic distributions.

A threshold value for PRCC of 0.25 is consistent with the guidance in NUREG/CR-6676 for derivation of site-specific values and Draft NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria," Appendix O, for derivation of site-specific distribution coefficients for soils.

The selection process takes into account the site-specific physical environment, importance of parameters, and the receptor's behavioral pattern and metabolic characteristics. The approach is consistent with NRC guidance and should result in the derivation of conservative site-specific DCGLs.

2.5.5.2.2 Dose Limit Reduction

In the development of site-specific DCGLs for the Industrial Area, the licensee decided to reduce the 25 mrem/yr dose criterion for unrestricted use per 10 CFR 20.1402. The licensee accounts for doses from discounted radionuclides of 0.054 mrem/yr, and from tritium contamination in three underlying ground water aquifer zones of 0.776 mrem/yr, resulting in a reported reduced dose limit of 24.18 mrem/yr. This dose limit reduction is incorrect based on the numbers reported, however, the error results in less than 0.5 percent difference in the licensee's reported DCGLs for the Industrial Area. The NRC staff reviewed the licensee's methodology and found it to be an acceptable approach. The following sections detail the staff's review of the licensee's methodology for determination of the dose limit reduction.

2.5.5.2.2.1 Discounted Radionuclides

Section 6.7.2 of the LTP describes the licensee's methodology for estimating the potential discarded dose from discounted radionuclides. Of the suite of 24 potential radionuclides of concern, they have observed only six in BRP environs during site characterization activities. The licensee uses RESRAD Version 6.21 to calculate the potential dose from discounted radionuclides based on the median minimum detectable activity for each of the discounted radionuclides from the laboratory analyses. The RESRAD model is prepared in the same manner used to derive site-specific input parameters, a review of which is above in Section 2.5.5.2.1. Furthermore, the licensee has only observed soil contamination in small areas no more than 150 m². Site characterization activities have not identified soil contamination above the characterization DCGLs at depths greater than 30 centimeters. Therefore, the licensee elects to calculate the discounted radionuclide potential dose using a contamination zone thickness of 0.3 meters and a contamination zone area of 150 square meters. Finally, the licensee selects the RESRAD estimated peak dose from all discounted radionuclides at four years, 0.054 mrem/yr, to reduce the dose limit for development of DCGLs for the Industrial Area. This is because the site characterization analyses occurred in 2001 and 2002 and site release will not occur before 2006. CE estimates that the peak dose occurs at this time or prior except for three radionuclides, Tc-99, I-129, and Pu-241. However, the peak dose estimated for each of these three radionuclides is less than 10 percent of the 25 mrem/yr dose limit. The peak dose for all three radionuclides occurs after the estimated peak doses from the six radionuclides observed in BRP environs and the three europium isotopes used to determine Industrial Area DCGLs. The NRC staff reviewed the licensee's methodology and finds it to be an acceptable approach.

2.5.5.2.2.2 Ground water Contamination

Section 6.8.2.1 of the LTP describes the licensee's methodology for estimating the dose contributions from tritium-contaminated ground water in three underlying aquifers. The licensee models the upper and intermediate ground water aquifers, independently, as contaminated zones with cover to determine peak doses from observed tritium contamination using RESRAD

Version 6.21. The licensee models these ground water zones in this manner because the ground water zones are not considered to be viable sources of potable water; this is because of the near-surface locations, limited thicknesses and pumping capacities. Furthermore, State of Michigan statutes prohibit drilling wells this shallow, and a more readily accessible source of water is nearby Lake Michigan. The licensee estimates peak doses from tritium in ground water of 0.024 mrem/yr and 0.395 mrem/yr for the upper and intermediate ground water aquifers, respectively. Furthermore, it estimates that these dose contributions occur within the first few years, which is well within the first 100 years following decommissioning. Therefore, this approach appears consistent with guidance in NUREG-1727, Appendix C.

Additionally, the licensee estimates the peak dose from tritium contamination in the bedrock aquifer by comparison with EPA's 40 CFR 141.16 drinking water standard. The licensee contends the bedrock aquifer is the most likely source of potable ground water. The licensee estimates a drinking water dose due to tritium contamination in the bedrock aquifer of 0.312 mrem/yr. The licensee increases this dose to account for the addition of the dose from the plant pathway through irrigation resulting in a total dose due to tritium contamination of the bedrock aquifer of 0.347 mrem/yr. The NRC staff's independent analysis indicates the licensee's estimated dose due to tritium contamination of the bedrock aquifer is likely conservative with respect to exposure to tritium through the drinking water pathway, which is expected to be the dominant exposure pathway for tritium.

The licensee sums the peak dose contributions due to tritium contamination of all three ground water aquifers to yield a total dose contribution of 0.776 mrem/yr. The licensee uses the estimated dose contributions from the tritium contamination of the three subsurface ground water aquifers combined with dose reduction from discounted radionuclides to reduce the 25 mrem/yr dose criterion when developing site-specific DCGLs for the Industrial Area. The NRC staff finds that the licensee's approach to reduce the DCGL limit based on estimated doses from tritium contamination of the three ground water aquifers and discounted radionuclides is acceptable.

2.5.5.2.3 Subsurface Soil Contamination

In Section 6.8.2.3 of the LTP, the licensee evaluates the potential effects of subsurface soil contamination (i.e., soil contamination found at depths greater than 0.15 meters). Site characterization activities have only located subsurface soil contamination in discrete pockets of small area and volume at depths above 1.5 meters. Therefore, the licensee contends that the dose limit reduction methodology discussed in the previous section is not appropriate for subsurface soil contamination. Instead, the licensee demonstrates that application of the site-specific DCGLs for the Industrial Area is appropriate down to depths of 10.7 meters; this coincides with the base of the containment structure and is considered by the licensee to be the depth to which a potential exists for subsurface soil contamination. Specifically, the licensee compares potential doses for the six radionuclides observed in BRP environs and the three europium isotopes from discrete pockets of 1.5 meters thick contamination zones at four depths (0.15, 1.5, 2.7, and 10.7 meters) with doses from a 0.15 meter thick contaminated zone at the surface. For the four discrete contamination assessments, the licensee maintains the site mathematical model, except the contamination area is fixed at 100 square meters – instead of 8094 square meters associated with the surface contamination assessment – to reflect the discrete pockets of contamination. Additionally, the licensee adjusts the unsaturated zone

properties in each of the four discrete contamination assessments to reflect the varying depth of the contamination zone.

The dose estimates, listed in Table 6-11 of the LTP, demonstrate that peak doses from individual radionuclides decrease with increasing depth from the surface for all the radionuclides except tritium, Fe-55, and Sr-90. Dose estimates for Fe-55 and Sr-90 increase with depth only for approximately the first 1.5 meters from the surface. For the three radionuclides for which dose increases with depth, the peak doses never exceed the 24.18 mrem/yr reduced dose limit. Furthermore, the licensee develops site-specific DCGLs, as described previously, using a contaminated zone thickness of 1.5 meters. Therefore, the reported site-specific DCGLs for Fe-55 and Sr-90 should account for this increase. Tritium peak doses continue to increase with increasing depth. However, the licensee accounted for subsurface tritium contamination in three underlying aquifers. The licensee reports tritium soil analyses indicate no detectable soil tritium at approximately 2.5 to 3.0 meters and 9.5 meters depths; they report 0.049 pCi/g soil tritium at approximately 4.5 meters depth. This is significantly lower than the calculated soil value obtained in the estimate of dose limit reduction described in the previous section. Therefore, the NRC staff finds the licensee's approach is acceptable.

2.5.6 Uncertainty and Variability

In Section 6.9 of the LTP, the licensee reports the findings of an evaluation to determine sensitivity of the parameters used in the BRP RESRAD dose model. For the six radionuclides observed in BRP environs and three europium isotopes, it ranked the sensitive parameters in the order of their sensitivity to the calculated peak mean dose. The licensee performs analyses for both uncorrelated parameters and correlated parameters. The parameter distributions used in the sensitivity analyses are the generic distributions from NUREG/CR-6697. Actual distributions for the BRP site are expected to be narrower because the generic distributions are based on national data. The licensee determines sensitive parameters based on values of PRCC calculated by the RESRAD code for each individual parameter. The PRCC is a gauge of the correlation between the peak radiation dose and the parameter value. Larger absolute values of the PRCC imply greater influence of the parameter value on the estimated peak dose. Positive PRCC values imply the estimated peak dose should increase with increases in the parameter value. Conversely, negative PRCC values imply the estimated peak dose should decrease with increases in the parameter value. Previous evaluations of uncertainty analyses published in NUREG/CR-6755, NUREG/CR-6697, NUREG/CR-6692, and NUREG/CR-6697 indicate that the PRCC is the most representative among several coefficients of correlation between the estimated peak dose and the parameter value.

2.5.7 Elevated Measurement Comparison DCGLs

In Section 6.10 of the LTP, the licensee proposes the methods it will use to derive the area factors associated with surface and subsurface soil. The licensee may use the area factors for elevated measurement comparisons to determine whether a smaller area of residual radioactivity exceeds the DCGLs during scanning in Industrial Area survey units. The licensee will need to adjust the number of static measurements if the sensitivity of the scanning technique is inadequate to detect levels of residual radioactivity below the DCGLs. Area factors

are also necessary to identify small areas with elevated residual radioactivity that may require further investigation.

The licensee calculates area factors for the modified resident farming scenario. The licensee calculated area factors using RESRAD Version 6.21 repeatedly with changing areas of contamination. Appendix 6-L of the LTP lists area factors for all six radionuclides observed in BRP environs and three europium isotopes. The NRC staff independently verified the licensee's calculated area factors using RESRAD Version 6.21 for the modified resident farmer scenario and finds no discrepancies.

2.6 Site End Use

Section 50.82(a)(9)(ii)(E) requires a licensee to describe the planned end use of the site if the licensee proposes to have its license terminated under restricted conditions. CE proposed to have its license terminated with no restrictions on the use of the site, under the provisions of 10 CFR 20.1402. Therefore, regulations do not require the licensee to describe the planned end use of the site.

The staff finds that the licensee conformed to 10 CFR 50.82(a)(9)(ii)(E) and the description is therefore acceptable.

2.7 Cost Estimate

The cost estimates included in Chapter 7 of the BRP LTP Rev. 0 were updated by letter from K. M. Haas to NRC, RE: Certification of Financial Assurance for Decommissioning, dated March 27, 2003. The modified estimates, in 2002 constant dollars, are total costs, \$395.3 million; site restoration, \$27.3 million; and spent fuel management, \$68.6 million. In accordance with NRC regulations, fuel storage and site restoration costs are not considered decommissioning costs. Therefore, the cost estimate for actual decommissioning of the facility is \$299.4 million, which includes an overall contingency of \$13 million, and \$44 million for radioactive waste disposal. As of December 21, 2002, the licensee has spent \$214.5 million, resulting in an estimated cost of \$84.9 million to complete decommissioning.

Many dismantle/decommission activities and tasks at BRP are similar to activities and tasks conducted at other nuclear generating plants that are undergoing decommissioning or have been decommissioned. Therefore, the staff's evaluation of BRP's decommissioning cost estimate was based on a comparison of activities to be conducted at BRP with the costs for similar activities conducted at other nuclear facilities that are undergoing decommissioning or have completed decommissioning. The staff's review included an evaluation of the licensee's cost assumptions used for estimating major decommissioning activities and tasks; it included a review of the dismantlement and decontamination costs, waste disposal costs, and FSS cost. It also included a comparison to similar activities conducted at Yankee Rowe Nuclear Power Station and Trojan Nuclear Power Plant. The staff recognizes that since the licensee initially developed the cost estimate, BRP has completed more than 50 percent of the decommissioning. Therefore, much of the cost information is no longer relevant.

The unit cost adjustment factors that were used in the licensee's cost estimate were similar to adjustment factors used at other facilities undergoing dismantlement. The staff also compared

the specific factors identified in Section 7.2 of the BRP LTP that the licensee used to develop the cost estimate, the cost factors in NUREG-1700, and the cost factors for developing the Trojan and Yankee Rowe estimates. The specific factors used for the BRP cost estimate compared favorably with the other facilities, and were reasonable.

The NRC staff reviewed the cost allocated for waste disposal (\$44 million), and the basis for the cost estimate. A large portion of the BRP waste is Class A waste, which they will send either to GTS Duratek for waste processing, or to the Envirocare facility. The staff concluded the cost estimate for waste disposal is reasonable.

The staff compared the BRP LTP with Section B.7 of NUREG-1700. Based on this review, the staff concludes that the licensee met the requirements of 10 CFR 50.82(a)(9)(ii)(F) by providing an updated site-specific cost estimate for the remaining decommissioning activities.

2.8 Environmental Report

NRC regulations require licensees to provide a supplement to the environmental report describing any new information or significant environmental changes associated with the licensee's proposed license termination activities. Chapter 8 of the LTP updates the "Environmental Report for Big Rock Point" (Ref. 22), submitted on September 18, 2002. Therefore, Chapter 8 of the LTP constitutes a supplement to BRP's Environmental Report, as required by 10 CFR 51.53(d) and 10 CFR 50.82(a)(9)(ii)(G). Based on the information in Chapter 8, the licensee concluded that the environmental impacts associated with changes in BRP's decommissioning activities remain bounded by the previously issued "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," NUREG-0856 (Ref. 23). Under the provisions of 10 CFR 51.21, the staff prepared an environmental assessment (EA) (Ref. 24) to determine the impacts of the proposed action on the environment. In the EA, the staff found that approval of the LTP would not cause any significant impacts on the human environment and is protective of human health.

The staff reviewed the information in the LTP for BRP, according to Section B.8 of NUREG-1700. Based on this review and the EA prepared by the staff, the staff concludes that the licensee met the regulatory requirements.

2.9 Change Procedure

The licensee proposed that it be authorized to make certain changes to the NRC-approved LTP without NRC approval. Once the LTP has been approved, the following change criteria will be used in addition to those criteria specified in 10 CFR 50.59, 10 CFR 72.48, 10 CFR 50.82(a)(6), and 10 CFR 50.82(a)(7). Changes to the LTP that require NRC approval prior to implementation include:

- Reducing the coverage requirements for scan measurements
- Using statistical tests other than the Sign Test for data evaluation
- Increasing the radionuclide-specific DCGLs or area factors
- Increasing the probability of making a Type I decision error above the level stated in the LTP
- Increasing the investigation level thresholds for a given survey unit classification

Any increase for any individual DCGL or collective increase of all DCGLs over all identified media is allowable only if the resulting total annual dose remains less than or equal to the specified dose limit.

CE will evaluate changes to the LTP using the DQO process outlined in MARSSIM and the considerations described in Section 1.6 of the LTP. They will submit changes to the LTP not requiring NRC approval as an update to the final safety analysis report, in accordance with 10 CFR 50.71(e).

Based on its evaluation of the LTP, the NRC staff concludes that authorizing the licensee to make certain changes to the LTP during the final site remediation is acceptable, subject to the above conditions.

2.10 Summary of Areas Requiring Further Validation

As a result of the staff's review of the LTP, the staff concludes that there are several areas related to the LTP that require validation either as part of the staff's ongoing inspection efforts of the BRP site or during the FSS :

- The staff concludes that the licensee identified the remaining dismantlement activities necessary to complete decommissioning of the facility as required by 10 CFR 50.82(a)(9)(ii)(B). Further, the staff concludes that these activities can be completed in accordance with 10 CFR 50.59. However, the staff will confirm, during in-process inspections, that the licensee conducts such activities in accordance with approved procedures and NRC regulations in Parts 20 and 50.
- The staff concludes that the approach the licensee used to characterize soils (surface and deep), concrete, pavement, sediments, embedded and buried piping, and surface and ground water is acceptable. The licensee committed to investigate further the presence and concentrations of plant-derived radionuclides, including HTDs and TRUs, in confirming that the assumptions used in developing the site dose model and cleanup criteria are still in agreement with actual site conditions. Accordingly, the staff will confirm, during in-process inspections, that the licensee implemented this commitment in confirming that residual contamination levels meet the release criteria. In addressing the selection of radionuclide distributions and establishing radionuclide fractions, NRC may collect specific samples and conduct independent radiochemical analyses in confirming results of the licensee's own analyses.
- The staff concludes that the approach proposed by the licensee to survey and characterize contamination as localized radioactivity levels is generally acceptable. The licensee committed to investigate further the presence of such contaminants on surfaces and soils and develop the necessary survey and sampling methods and sample analysis. Accordingly, the staff will confirm, during in-process inspections and side-by-side confirmatory surveys, that the proposed survey procedure is adequate in identifying localized areas of elevated levels of activity and that remediation was effective in removing such contaminants, before implementing FSS.

- The licensee proposes the possible use of *in situ* gamma spectrometry and other technologies under certain conditions (LTP Section 5.4.3). However, minimal details are included in the LTP at this time. The licensee states that they will provide a technical basis document to NRC for review once such a need has been identified and the appropriate survey procedure has been developed. The staff will confirm, during in-process inspections, that the proposed survey method and procedures are adequate in confirming that areas surveyed by this method meet the site release criteria.
- The licensee committed to install ground water wells, to replace those that they will demolish during remediation, for monitoring ground water quality. In addressing the installation of replacement wells, the staff finds the proposed commitment acceptable, but will require evaluation of the basis for the location and number of replacement wells. Accordingly, the NRC staff will evaluate, during in-process inspections, the licensee's justification for replacing specific wells. In addition, the staff will conduct independent sampling and analysis of ground water samples, to confirm that the licensee met the ground water concentration limits of Section 6.0, Table 6-11, of the LTP.

3.0 STATE CONSULTATION

In accordance with NRC regulations, NRC notified the State of Michigan of the proposed issuance of the amendment. The Michigan Department of Environmental Quality submitted comments on the EA on September 2, 2004. Briefly, the State did not agree that all non-radiological remediation was complete; it is also aware of the licensee's actions to dispose of non-radiological waste. NRC staff responded to the State's comments in the revised EA.

4.0 ENVIRONMENTAL CONSIDERATIONS

The objective of decommissioning the BRP site is to reduce the residual radioactivity to levels that permit release of the site for unrestricted use and for termination of the 10 CFR Part 50 license, in accordance with the site release criteria in 10 CFR Part 20. The purpose of the LTP is to satisfy the requirements of 10 CFR 50.82, "Termination of license." The LTP describes the decommissioning activities to be performed by the licensee, the process for performing the final status surveys, and the methods for demonstrating that the site meets the criteria for release for unrestricted use. Therefore, the LTP involves the reduction of radioactivity at the site, and does not involve in any way the operation of the plant and the generation of new radioactive material. Based on this and on the evaluation given above on the proposed amendment to approve the LTP, this amendment incorporates into the BRP LTP the LTP change process, which allows the licensee to make changes to the plan without NRC review and approval. Pursuant to 10 CFR 51.21, 51.32, and 51.35, an EA and Finding of No Significant Impact were published in the *Federal Register* on March 21, 2005.

Based on the EA, the NRC determined that issuing this amendment will not have a significant effect on the quality of the human environment. Accordingly, a Finding of No Significant Impact is appropriate.

5.0 CONCLUSIONS

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that operation in the proposed manner will not endanger the health and safety of the public; (2) the licensee will conduct such activities in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 LIST OF CONTRIBUTORS

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7.0 LIST OF ACRONYMS

ALARA	As Low As Is Reasonable Achievable
BWR	Boiling Water Reactor
BRP	Big Rock Point
Bq/g	Becquerel per gram
Bq/L	Becquerel per liter
CFR	<u>Code of Federal Regulations</u>
DCGL	Derived Concentration Guideline Limit
DOE	U.S. Department of Energy
dpm/100cm ²	disintegrations per minute per 100 square centimeters
DQO	Data Quality Objective
EA	Environmental Assessment
FR	<u>Federal Register</u>
FSS	Final Status Survey
HSA	Historical Site Assessment
HTD	Hard to Detect
IA	Industrial Area
IM	Impacted Area
ISFSI	Independent Spent Fuel Storage Installation
LBGR	Lower Boundary of the Gray Region
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey And Site Investigation Manual
MDA	Minimum Detection Activity
MDC	Minimum Detectable Concentration
mrem/hr	millirem per hour
mrem/yr	millirem per year
MSL	Mean Sea Level
mSv/yr	milliSievert per year
NIST	National Institute of Standards and Technology
NMSS	Office of Nuclear Material Safety and Safeguards

NRC	Nuclear Regulatory Commission
ORISE	The Oak Ridge Institute for Science and Education
pCi/g	picocurie per gram
pCi/L	picocurie per Liter
PRCC	Partial Rank Correlation Coefficient
QA	Quality Assurance
QC	Quality Control
RAI	Request for Additional Information
RCRA	Resource Conservation and Recovery Act
SER	Safety Evaluation Report
SRP	Standard Review Plan
TEDE	Total Effective Dose Equivalent
TLG	TLG Services
TRU	Transuranic
uR/hr	microrentgen per hour

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